UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 27, 1994

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M86095, M86096, AND M86097) (TS 322)

The Commission has issued the enclosed Amendment Nos. 212, 227, and 185 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated April 4, 1994, regarding the reactor scram and system isolation functions generated by the Main Steamline Radiation Monitors (MSRMs). That application superseded an earlier amendment request on this subject dated March 25, 1993.

The amendments eliminate the requirements in the Technical Specifications (TS) for automatic actuation of the following functions upon MSRM detection of a high radiation condition in the main steam lines:

(1) reactor scram
(2) main steam isolation valve closure
(3) main steam line drain valve closure
(4) reactor recirculation sample line valve closure
(5) main condenser mechanical vacuum pump isolation and trip

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AMENDMENT NO. 212 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 227 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 185 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: September 27, 1994

<u>Distribution w/enclosure</u>

Docket File	
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A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Joseph F. Williams

Joseph F. Williams, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosures:
1. Amendment No. 212 to License No. DPR-33
2. Amendment No. 227 to License No. DPR-52
3. Amendment No. 185 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED:

Joseph F. Williams, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc w/enclosures: See next page

DOCUMENT NAME: G:\BFN\TS322.AMD *SEE PREVIOUS CONCURRENCE
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Tennessee Valley Authority

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 212, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 212

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* pages are provided to maintain document completeness.

REMOVE	INSERT
3.1/4.1-3	3.1/4.1-3*
3.1/4.1-4	3.1/4.1-4
3.1/4.1-5	3.1/4.1-5*
3.1/4.1-6	3.1/4.1-6
3.1/4.1-8	3.1/4.1-8*
3.1/4.1-9	3.1/4.1-9
3.1/4.1-11	3.1/4.1-11
3.1/4.1-12	3.1/4.1-12
3.1/4.1-14	3.1/4.1-14*
3.1/4.1-15	3.1/4.1-15
3.2/4.2-7	3.2/4.2-7*
3.2/4.2-8	3.2/4.2-8
3.2/4.2-12	3.2/4.2-12*
3.2/4.2-13	3.2/4.2-13
3.2/4.2-40	3.2/4.2-40
3.2/4.2-41	3.2/4.2-41*
3.2/4.2-67	3.2/4.2-67
3.2/4.2-68	3.2/4.2-68*
3.7/4.7-33	3.7/4.7-33*
3.7/4.7-34	3.7/4.7-34
3.8/4.8-3	3.8/4.8-3*
3.8/4.8-4	3.8/4.8-4
3.8/4.8-9	3.8/4.8-9
3.8/4.8-10	3.8/4.8-10*

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X(21)	X(17)	(15)	1.A
2	Inoperative	(13)		X(21)	X(17)	X	1.A
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable: Refuel (7)	Startup/ Hot Standby	Run	Action (1)
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure		X(3)(6)	X(3)(6)	X(6)	1.A or 1.C
2	Turbine Control Valve Fast Closure or Turbine Trip	≥550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	not ≥154 psig		X(18)	X(18)	X(18)	1.A or 1.D (19)

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for one trip system, trip the inoperable channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of operable instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An inoperable channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable channel shall be restored to operable status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all operable control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. Bypassed if reactor pressure is less than 1055 psig and mode switch not in RUN.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15 percent scram

NOTES FOR TABLE 3.1.A (Cont'd)

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first state pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5×10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

TABLE 4.1.A (Continued)

	Group (2)	Functional Test	Minimum Frequency(3)
High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
Electronic Level Switches (LS-85-45A, B, G, H)	A	Trip Channel and Alarm	Once/Month
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/3 Months (8)
Turbine Control Valve Fast Closure or turbine trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive (PT-1-81A and B, PT-1-91A and B)	B	Trip Channel and Alarm (7)	Every three months
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/Month (1)

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Electronic Lvl Switches (LS-85-45-A, B, G, H)	A	Calibrated Water Column (5)	Note (5)
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PT-1-81A, B & PT-1-91A, B)	B	Standard Pressure Source	Once/Operating Cycle (9)
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

3.1 BASES (Cont'd)

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation (Groups 2, 3, and 6) c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and 94)	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 378" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Low Pressure Main Steam Line	$\geq$ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	$\leq$ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
2(14)	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1(15)	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

NOTES FOR TABLE 3.2.A

1. Whenever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

   E. Initiate primary containment isolation within 24 hours.

   F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

   G. Isolate the reactor building and start the standby gas treatment system.

   H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

   I. Deleted

   J. Withdraw TIP.

   K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

   L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.

9. Deleted

10. Deleted

11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. For the Reactor Building Ventilation system, one channel may be inoperable for up to 4 hours for functional testing or for up to 24 hours for calibration and maintenance, as long as the downscale trip of the inoperable channel is placed in the tripped condition.

12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

    Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

    In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. Deleted

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D, SW 2-3)	(1)	(5)	once/day
Instrument Channel - Reactor High Pressure (PS-68-93 & -94)	(31)	once/18 months	None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	(1)	once/3 month	once/day
Instrument Channel - High Drywell Pressure (PS-64-56A-D)	(1)	(5)	N/A
Instrument Channel - Low Pressure Main Steam Line (PT-1-72, -76, -82, -86)	once/3 months (27) (29)	once/operating cycle (28)	None
Instrument Channel - High Flow Main Steam Line (dPT-1-13A-D, -25A-D, -36A-D, -50A-D)	once/3 months (27) (29)	once/operating cycle (28)	once/day

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Main Steam Line Tunnel High Temperature	once/3 months (27)	once/operating cycle	None
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1) (30)	once/18 months	once/day (8)
Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	(1) (30)	once/18 Months	once/day (8)
Instrument Channel - SGTS Train A Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train B Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train C Heaters	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

3.2 BASES (Cont'd)

The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and OPERABLE during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.7/4.7 BASES (Cont'd)

Demonstration of the automatic initiation capability and OPERABILITY of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the OPERABLE systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifictions. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at 378".

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

LIMITING CONDITIONS FOR OPERATION

3.8.B. Airborne Effluents

1. (Deleted)

2. (Deleted)

3. (Deleted)

4. (Deleted)

5. (Deleted)

6. (Deleted)

7. (Deleted)

8. (Deleted)

9. Whenever the SJAE is in service, the concentration of hydrogen in the offgas downstream of the recombiners shall be limited to ≤4% by volume.

10. With the concentration of hydrogen exceeding the limit of 3.8.B.9 above, restore the concentration to within the limit within 48 hours.

SURVEILLANCE REQUIREMENTS

4.8.B. Airborne Effluents

1. (Deleted)

2. (Deleted)

3. (Deleted)

4. (Deleted)

5. The concentration of hydrogen downstream of the recombiners shall be determined to be within the limits of 3.8.B.9 by continuously monitoring the off-gas whenever the SJAE is in service using instruments described in Table 3.2.K. Instrument surveillance requirements are specified in Table 4.2.K.

3.8/4.8 RADIOACTIVE MATERIALS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C (Deleted)	4.8.C (Deleted)
3.8.D (Deleted)	4.8.D (Deleted)

3.8 BASES

(Deleted)

3.8.A LIQUID HOLDUP TANKS

Specification 3.8.A.5 includes any tanks containing radioactive material that are not surrounded by liners, dikes, or walls capable of holding the contents and that do not have overflows and surrounding area drains connected to the liquid radwaste treatment system. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3.8.B EXPLOSIVE GAS MIXTURE

Specification 3.8.B.9 and 10 is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas system is maintained below the flammability limits of hydrogen. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

4.8.A and 4.8.B BASES

(Deleted)

3.8.C and 4.8.C BASES

(Deleted)

3.8.D and 4.8.D BASES

(Deleted)

3.8.E and 4.8.E BASES

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 227
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 227, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* pages are provided to maintain document completeness.

REMOVE	INSERT
3.1/4.1-3	3.1/4.1-3*
3.1/4.1-4	3.1/4.1-4
3.1/4.1-5	3.1/4.1-5*
3.1/4.1-6	3.1/4.1-6
3.1/4.1-8	3.1/4.1-8*
3.1/4.1-9	3.1/4.1-9
3.1/4.1-11	3.1/4.1-11
3.1/4.1-11a	-----
3.1/4.1-12	3.1/4.1-12
3.1/4.1-14	3.1/4.1-14*
3.1/4.1-15	3.1/4.1-15
3.2/4.2-7	3.2/4.2-7*
3.2/4.2-8	3.2/4.2-8
3.2/4.2-12	3.2/4.2-12*
3.2/4.2-13	3.2/4.2-13
3.2/4.2-40	3.2/4.2-40
3.2/4.2-41	3.2/4.2-41*
3.2/4.2-67	3.2/4.2-67
3.2/4.2-68	3.2/4.2-68*
3.7/4.7-33	3.7/4.7-33*
3.7/4.7-34	3.7/4.7-34
3.8/4.8-3	3.8/4.8-3*
3.8/4.8-4	3.8/4.8-4
3.8/4.8-9	3.8/4.8-9
3.8/4.8-10	3.8/4.8-10*

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X(21)	X(17)	(15)	1.A
2	Inoperative	(13)		X(21)	X(17)	X	1.A
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure (PIS-3-22AA,BB,C,D)	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56 A-D)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203 A-D)	≥ 538" above vessel zero		X	X	X	1.A

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure				X(6)	1.A or 1.C
2	Turbine Control Valve Fast Closure or Turbine Trip	≥550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B)	not ≥154 psig		X(18)	X(18)	X(18)	1.A or 1.D (19)
2	Low Scram Pilot Air Header Pressure	≥50 psig	X(2)	X(2)	X	X	1.A

NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

2. Scram discharge volume high bypass may be used in SHUTDOWN or REFUEL to bypass scram discharge volume scram and scram pilot air header low pressure scram with control rod block for reactor protection system reset.

3. (Deleted)

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in SHUTDOWN

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15 percent scram

   F. Scram pilot air header low pressure

NOTES FOR TABLE 3.1.A (Cont'd)

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5 x 10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Trip Channel and Alarm (7)	Once/Month
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and Alarm (7)	Once/Month
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (7)	Once/Month

TABLE 4.1.A (Continued)

	Group (2)	Functional Test	Minimum Frequency(3)
High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/Month
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/3 Months (8)
Turbine Control Valve Fast Closure or turbine trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive (PIS-1-81A and B, PIS-1-91A and B)	B	Trip Channel and Alarm (7)	Every three months
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/Month (1)
Low Scram Pilot Air Header Pressure (PS 85-35 A1, A2, B1, & B2)	A	Trip Channel and Alarm	Once/6 Months

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22 AA, BB, C, D)	B	Standard Pressure Source	Once/6 Months (9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/18 Months (9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/18 Months (9)
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45-C-F)	A	Calibrated Water Column	Once/18 Months
Electronic Level Switches (LS-85-45 A, B, G, H)	B	Calibrated Water Column	Once/18 Months (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Standard Pressure Source	Once/18 Months (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Turbine Control Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Low Scram Pilot Air Header Pressure (PS 85-35 A1, A2, B1, & B2)	A	Standard Pressure Source	Once/18 Months

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

3.1 BASES (Cont'd)

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6) (LIS-3-203 A-D)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and -94)	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	≥ 398" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	$\geq$ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	$\leq$ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
1(14)	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

NOTES FOR TABLE 3.2.A

1. Whenever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

   E. Initiate primary containment isolation within 24 hours.

   F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

   G. Isolate the reactor building and start the standby gas treatment system.

   H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

   I. Deleted

   J. Withdraw TIP.

   K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

   L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.

9. Deleted

10. Deleted

11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. For the Reactor Building Ventilation system, one channel may be inoperable for up to 4 hours for functional testing or for up to 24 hours for calibration and maintenance, as long as the downscale trip of the inoperable channel is placed in the tripped condition.

12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

    Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

    In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. Deleted

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency		Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel - Reactor High Pressure (PS-68-93 & 94)	(31)	Once/18 months		None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	(1) (27)	Once/18 months	(28)	Once/day
Instrument Channel - High Drywell Pressure (PIS-64-56A-D)	(1) (27)	Once/18 Months	(28)	N/A
Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(29) (27)	Once/18 Months (28)		None
Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	(29) (27)	Once/18 Months (28)		Once/day

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Main Steam Line Tunnel High Temperature	once/3 months (27)	once/operating cycle	None
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1) (32)	once/18 months	once/day (8)
Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	(1) (32)	once/18 Months	once/day (8)
Instrument Channel - SGTS Train A Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train B Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train C Heaters	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. Each trip system consists of two elements. Each channel contains one temperature switch located in the pump room and three temperature switches located in the torus area. The RCIC high flow and high area temperature sensing instrument channels are arranged in the same manner as the HPCI system.

The HPCI high steam flow trip setting of 90 psid and the RCIC high steam flow trip setting of 450" H_2O have been selected such that the trip setting is high enough to prevent spurious tripping during pump startup but low enough to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

The HPCI and RCIC steam line space temperature switch trip settings are high enough to prevent spurious isolation due to normal temperature excursions in the vicinity of the steam supply piping. Additionally, these trip settings ensure that the primary containment isolation steam supply valves isolate a break within an acceptable time period to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

High temperature at the Reactor Water Cleanup (RWCU) System in the main steam valve vault, RWCU pump room 2A, RWCU pump room 2B, RWCU heat exchanger room or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

A General Electric study, GE-NE-770-06-0392 shows for the unit 2 cycle 6 core that if the initial MCPR is as specified in item 7e or 7f of Table 3.2.C, then no single rod withdrawal error can cause the MCPR to decrease below the MCPR safety limit. When core operating conditions have been verified to be within the limits of items 7e or 7f of Table 3.2.C, the RBM is not required. When the RBM is required, the minimum instrument channel requirements apply. These requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and OPERABLE during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and OPERABILITY of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the OPERABLE systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level ($\geq$ 398") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at $\geq$ 398".

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

LIMITING CONDITIONS FOR OPERATION

3.8.B. Airborne Effluents

1. (Deleted)
2. (Deleted)
3. (Deleted)
4. (Deleted)
5. (Deleted)
6. (Deleted)
7. (Deleted)
8. (Deleted)
9. Whenever the SJAE is in service, the concentration of hydrogen in the offgas downstream of the recombiners shall be limited to ≤4% by volume.
10. With the concentration of hydrogen exceeding the limit of 3.8.B.9 above, restore the concentration to within the limit within 48 hours.

SURVEILLANCE REQUIREMENTS

4.8.B. Airborne Effluents

1. (Deleted)
2. (Deleted)
3. (Deleted)
4. (Deleted)
5. The concentration of hydrogen downstream of the recombiners shall be determined to be within the limits of 3.8.B.9 by continuously monitoring the off-gas whenever the SJAE is in service using instruments described in Table 3.2.K. Instrument surveillance requirements are specified in Table 4.2.K.

3.8/4.8 RADIOACTIVE MATERIALS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C (Deleted)	4.8.C (Deleted)
3.8.D (Deleted)	4.8.D (Deleted)

3.8 BASES

(Deleted)

3.8.A LIQUID HOLDUP TANKS

Specification 3.8.A.5 includes any tanks containing radioactive material that are not surrounded by liners, dikes, or walls capable of holding the contents and that do not have overflows and surrounding area drains connected to the liquid radwaste treatment system. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3.8.B EXPLOSIVE GAS MIXTURE

Specification 3.8.B.9 and 10 is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas system is maintained below the flammability limits of hydrogen. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

4.8.A and 4.8.B BASES

(Deleted)

3.8.C and 4.8.C BASES

(Deleted)

3.8.D and 4.8.D BASES

(Deleted)

3.8.E and 4.8.E BASES

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* pages are provided to maintain document completeness.

REMOVE	INSERT
3.1/4.1-2	3.1/4.1-2*
3.1/4.1-3	3.1/4.1-3
3.1/4.1-4	3.1/4.1-4*
3.1/4.1-5	3.1/4.1-5
3.1/4.1-7	3.1/4.1-7*
3.1/4.1-8	3.1/4.1-8
3.1/4.1-10	3.1/4.1-10
3.1/4.1-11	3.1/4.1-11
3.1/4.1-13	3.1/4.1-13*
3.1/4.1-14	3.1/4.1-14
3.2/4.2-7	3.2/4.2-7*
3.2/4.2-8	3.2/4.2-8
3.2/4.2-12	3.2/4.2-12*
3.2/4.2-13	3.2/4.2-13
3.2/4.2-39	3.2/4.2-39
3.2/4.2-40	3.2/4.2-40*
3.2/4.2-66	3.2/4.2-66
3.2/4.2-67	3.2/4.2-67*
3.7/4.7-32	3.7/4.7-32*
3.7/4.7-33	3.7/4.7-33
3.8/4.8-3	3.8/4.8-3*
3.8/4.8-4	3.8/4.8-4
3.8/4.8-9	3.8/4.8-9
3.8/4.8-10	3.8/4.8-10*

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X(21)	X(17)	(15)	1.A
2	Inoperative	(13)		X(21)	X(17)	X	1.A
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable				Action (1)
			Shut-down	Refuel (7)	Startup/ Hot Standby	Run	
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure				X(6)	1.A or 1.C
2	Turbine Control Valve Fast Closure or Turbine Trip	≥550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	not ≥154 psig		X(18)	X(18)	X(18)	1.A or 1.D (19)

NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT STANDBY position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. DELETED

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15 percent scram

NOTES FOR TABLE 3.1.A (Cont'd)

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5 x 10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

TABLE 4.1.A (Continued)

	Group (2)	Functional Test	Minimum Frequency(3)
High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/Month
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/3 Months (8)
Turbine Control Valve Fast Closure or turbine trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every three months
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/Month (1)

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Electronic Lvl Switches (LS-85-45-A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the Bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

3.1 BASES (Cont'd)

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation (Groups 2, 3, and 6) c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and 94)	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 378" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Low Pressure Main Steam Line	$\geq$ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	$\leq$ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
2(14)	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1(15)	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

NOTES FOR TABLE 3.2.A

1. Whenever the respective functions are required to be OPERABLE, there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in COLD SHUTDOWN CONDITION in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Administratively control the affected system isolation valves in the closed position within one hour and then declare the affected system inoperable.

   E. Initiate primary containment isolation within 24 hours.

   F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

   G. Isolate the reactor building and start the standby gas treatment system.

   H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

   I. DELETED

   J. Withdraw TIP.

   K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

   L. If one SGTS train is inoperable take action H or actions A and F. If two SGTS trains are inoperable take actions A and F.

2. Deleted

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

NOTES FOR TABLE 3.2.A (Cont'd)

4. Only required in RUN MODE (interlocked with Mode Switch).

5. Deleted

6. Channel shared by RPS and Primary Containment & Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require actions A and F.

9. Deleted

10. Refer to Table 3.7.A and its notes for a listing of Isolation Valve Groups and their initiating signals.

11. A channel may be placed in an inoperable status for up to four hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. For the Reactor Building Ventilation system, one channel may be inoperable for up to 4 hours for functional testing or for up to 24 hours for calibration and maintenance, as long as the downscale trip of the inoperable channel is placed in the tripped condition.

12. A channel contains four sensors, all of which must be OPERABLE for the channel to be OPERABLE.

    Power operations permitted for up to 30 days with 15 of the 16 temperature switches OPERABLE.

    In the event that normal ventilation is unavailable in the main steam line tunnel, the high temperature channels may be bypassed for a period of not to exceed four hours. During periods when normal ventilation is not available, such as during the performance of secondary containment leak rate tests, the control room indicators of the affected space temperatures shall be monitored for indications of small steam leaks. In the event of rapid increases in temperature (indicative of steam line break), the operator shall promptly close the main steam line isolation valves.

13. Deleted

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel – Reactor Low Water Level (LIS-3-203A-D, SW 2-3)	(1)	(5)	once/day
Instrument Channel – Reactor High Pressure (PS-68-93 & -94)	(31)	once/18 months	None
Instrument Channel – Reactor Low Water Level (LIS-3-56A-D, SW #1)	(1)	once/3 month	once/day
Instrument Channel – High Drywell Pressure (PS-64-56A-D)	(1)	(5)	N/A
Instrument Channel – Low Pressure Main Steam Line	once/3 months (27)	once/3 months	None
Instrument Channel – High Flow Main Steam Line	once/3 months (27)	once/3 months	once/day

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Main Steam Line Tunnel High Temperature	once/3 months (27)	once/operating cycle	None
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1) (30)	once/18 months	once/day (8)
Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	(1) (30)	once/18 Months	once/day (8)
Instrument Channel - SGTS Train A Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train B Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train C Heaters	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

3.2 BASES (Cont'd)

The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and OPERABLE during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and OPERABILITY of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the OPERABLE systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level (378") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at 378".

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.B. Airborne Effluents	4.8.B. Airborne Effluents
1. (Deleted)	1. (Deleted)
2. (Deleted)	2. (Deleted)
3. (Deleted)	3. (Deleted)
4. (Deleted)	4. (Deleted)
5. (Deleted)	5. The concentration of hydrogen downstream of the recombiners shall be determined to be within the limits of 3.8.B.9 by continuously monitoring the off-gas whenever the SJAE is in service using instruments described in Table 3.2.K. Instrument surveillance requirements are specified in Table 4.2.K.
6. (Deleted)	
7. (Deleted)	
8. (Deleted)	
9. Whenever the SJAE is in service, the concentration of hydrogen in the offgas downstream of the recombiners shall be limited to $\leq 4\%$ by volume.	
10. With the concentration of hydrogen exceeding the limit of 3.8.B.9 above, restore the concentration to within the limit within 48 hours.	

3.8/4.8 RADIOACTIVE MATERIALS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C (Deleted)	4.8.C (Deleted)
3.8.D (Deleted)	4.8.D (Deleted)

3.8 BASES

(Deleted)

3.8.A LIQUID HOLDUP TANKS

Specification 3.8.A.5 includes any tanks containing radioactive material that are not surrounded by liners, dikes, or walls capable of holding the contents and that do not have overflows and surrounding area drains connected to the liquid radwaste treatment system. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3.8.B EXPLOSIVE GAS MIXTURE

Specification 3.8.B.9 and 10 is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas system is maintained below the flammability limits of hydrogen. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

4.8.A and 4.8.B BASES

(Deleted)

3.8.C and 4.8.C BASES

(Deleted)

3.8.D and 4.8.D BASES

(Deleted)

3.8.E and 4.8.E BASES

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated March 25, 1993, the Tennessee Valley Authority (TVA) submitted a request for an amendment to Browns Ferry Nuclear Plant, Units 1, 2 and 3, Appendix A of Operating Licenses No. DPR-33, DPR-52 and DPR-68 in accordance with 10 CFR 50.90 (Reference 1). On April 4, 1994, TVA submitted a new amendment request that superseded the March 25, 1993, submittal (Reference 2). In the April 4, 1994, letter, TVA proposed Technical Specification (TS) changes for eliminating the main steamline high radiation monitor (MSRM) signal from initiating the following: 1) reactor scram, 2) main steamline isolation valve closure, 3) main steamline drain valves closure, and 4) reactor recirculation sample line valve closure. TVA stated that these changes will reduce scram frequency, maintain availability of the condenser heat sink, eliminate the potential for trips due to hydrogen water chemistry, and increase operator control over radioactive releases.

TVA also stated that the MSRM functions are not required to ensure compliance with the offsite radiation release guidelines of 10 CFR Part 100.

2.0 BACKGROUND

The MSRMs provide an early indication of gross fuel failure. When a high radiation level equal to 3 times the normal background level for full power is detected, a reactor scram is initiated to reduce the continued failure of fuel cladding. This same high radiation condition also signals the primary containment isolation system (PCIS) to initiate containment isolation to prevent the release of fission products. The main steamline isolation valves (MSIVs) are part of the PCIS and are signaled to close when a high level of radiation is detected by the MSRMs. The reactor scram and PCIS trip setting is high enough above normal background radiation levels to prevent spurious trips yet low enough to promptly detect gross failure in the fuel cladding. The MSRM alarm setpoint is 1.5 times background based on normal full power background radiation levels.

ENCLOSURE 4

3.0 EVALUATION

TVA proposed an amendment for Browns Ferry Nuclear Plant Units 1, 2 and 3 to revise TS Sections 3.1/4.1, 3.2/4.2, 3.7/4.7, and 3.8/4.8 by eliminating the main steamline high radiation signal as detected by the MSRM from initiating the following: 1) reactor scram, 2) MSIV closure, 3) main steamline drain valves closure, and 4) reactor recirculation sample line valve closure. The main condenser mechanical vacuum pump isolation and trip function will remain, but is eliminated from the TS. Corresponding changes to the TS Bases are included in the amendment.

The proposed change also eliminates the TS requirements for MSRM calibration and functional testing.

The MSRM alarm function will be maintained with a setpoint of 1.5 times normal full power background. This alarm function along with indication from other radiation monitors provides information to the operator for taking corrective action as necessary to reduce radiation activity releases or shut down the plant.

The above proposed changes are in accordance with the General Electric Topical Report, NEDO-31400, "Safety Evaluation for Eliminating The Boiling Water Reactor Main Steamline Radiation Monitor," dated May 1987, prepared for the Boiling Water Reactor Owners's Group (BWROG). The purpose of this report was to demonstrate that the reactor vessel isolation function and scram function of the MSRM are not required to ensure compliance with the radiation dose guidelines of 10 CFR Part 100 for a control rod drop accident (CRDA). In addition, the report demonstrated that use of the offgas treatment system provides significant holdup time for radionuclides, and is an acceptable method of controlling unexpected radioactivity releases. This report concluded that elimination of the MSIV closure function and scram function of the MSRM, in conjunction with proper use of an augmented offgas system, results in offsite radiological exposures that are a small fraction of 10 CFR Part 100 guidelines, even when utilizing very conservative source terms (Reference 3).

The staff Safety Evaluation Report (SER) dated May 15, 1991 (Reference 10) accepted NEDO-31400 for use as a reference in licensee applications to eliminate the MSRM reactor scram and MSIV closure, provided that:

1. Assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the specific plant.

2. There is sufficient evidence (such as implemented or proposed operating procedures or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steamlines would be controlled expeditiously to limit both occupational doses and environmental releases.

3. The MSRM and offgas radiation monitor alarm setpoints are set to 1.5 times the normal N-16 background dose rate at the monitoring locations, and commitments are made to promptly sample the reactor coolant for contamination levels if the MSRM or offgas radiation monitors, or both, exceed their alarm setpoints.

The licensee responded to the above conditions as follows:

R1 A comparison was performed by the licensee of the key analysis input values in NEDO-31400 to the values committed to in the Browns Ferry FSAR. The input values compared were: power (mw/rod), number of failed fuel rods, length of operation, normal and meltdown releases, CHI/Q at ground and elevated, and holdup (delay time). The licensee determined that the assumptions made in the generic analysis exceeded those values committed to in the FSAR; therefore, they bound the Browns Ferry licensing basis.

R2 The licensee stated that they have procedures in place which address the operator actions required in the event of high radiation in the main steam line. The staff requested that the licensee identify these procedures and send copies of those pages that address the applicable operator actions. TVA submitted the alarm response procedure (ARP) for Unit 2 for the main steam radiation high alarm which includes operator action (Reference 11) and the surveillance instruction (SI) for monitoring airborne effluent release rate (Reference 12). The SI was revised to include radiation release requirements removed from the TS and placed in the Offsite Dose Calculation Manual (ODCM). TVA also submitted chemical instructions (CI) for determining fuel performance and isotopic trends which provide information on fuel integrity (Reference 13).

The staff's review of the above ARP determined that the ARP makes reference to the above SI, other offgas and stack gas radiation recorders, and the wide range gaseous effluent radiation monitor, and directs the operators to take appropriate action to reduce activity or shut down the plant as necessary.

R3 The MSRM will be set to alarm at 1.5 times normal full-power background which includes the nitrogen-16 contribution. Browns Ferry has procedures for controlling the offgas monitor setpoints as part of the ODCM which implements 10 CFR Part 50, Appendix I requirements. An ARP requires prompt sampling of the reactor coolant to determine possible contamination levels and the need for additional corrective action if the MSRM or offgas radiation monitors or both exceed their alarm setpoints.

Elimination of the following MSRM functions are not covered by NEDO-31400: 1) main steamline drain valve (MSLDV) closure, 2) reactor recirculation sample line (RRSL) isolation, and 3) mechanical vacuum pump (MVP) trip and isolation. These functions are discussed below.

MSLDV CLOSURE

The three-inch main steamline drain header at Browns Ferry discharges directly to the condenser. Since the NEDO-31400 analysis and the additional Browns Ferry specific offsite dose calculation is based upon the entire CRDA source term being instantaneously deposited into the condenser via the 24" main steamlines, the main steamline drains cannot increase the source term in the condenser nor create an additional release path if they are not isolated. Therefore, their closure requirement on receipt of a MSRM signal can be eliminated from the TS.

RRSL CLOSURE

The RRSL currently receives a primary containment isolation signal (PCIS) for the following conditions: 1) low-low-low reactor water level; 2) MSRM high radiation; 3) high main steamline flow; 4) high main steamline temperature; and 5) low main steamline pressure. TVA provided an analysis for removing the MSRM signal from the RRSL isolation following a CRDA as follows.

The 3 RRSLs are connected to the discharge of the reactor recirculation pumps and are normally closed, unless the normal sample paths from the reactor water cleanup (RWCU) demineralizers are out of service. The RRSLs are connected to the same sample station as the RWCU system. The sample station is protected from over-pressurization by non-safety related over pressure protection devices. If the non-safety related over pressure protection devices fail following a CRDA, over-pressure of the sample station piping or instruments can occur producing a continuous blowdown of reactor coolant into the reactor building. Although TVA considers this scenario very improbable, an analysis was performed of the consequences of such an event. This analysis showed that the fission products released to the reactor building through a RRSL break following a CRDA would initiate isolation of secondary containment, and would start the standby gas treatment system (SGTS) on high radiation in the reactor building exhaust ducts. The SGTS releases are modeled from the plant stack and the resulting offsite doses from the above postulated RRSL release path are well below the 10 CFR 100 limits. Based on this analysis, the licensee stated that the RRSL isolation function of the MSRM is not required to mitigate a CRDA and can, therefore, be eliminated from the plant TS.

MVP TRIP AND ISOLATION

The NEDO-31400 CRDA analysis assumed the MSIVs did not isolate, but did assume a MVP trip and isolation; therefore, the MSRM initiation of MVP trip and isolation will remain functional. However, this function does not meet any of the four criteria for inclusion in the TS as described in the NRC "Final Policy Statement on Technical Specifications" (Reference 13). The MVP isolation and trip function will be described in plant procedures controlled by 10 CFR 50.59 change process. TVA has performed an additional conservative analysis of offsite releases using the same assumptions and input parameters as those in NEDO-31400, except that the MVP is assumed to continue to operate, rather than to trip and isolate. This analysis showed that the resulting offsite doses are still within the required 10 CFR Part 100 limits, and therefore, the MVP trip and isolation on receipt of a signal from the MSRM can be removed from the TS.

The staff has reviewed the results of the various analyses for offsite dose releases from elimination of the MSRM high radiation signal for reactor scram, MSIV closure, MSLDV closure, RRSL valve closure and MVP trip and isolation from the TS. Based on that review, the staff concludes that the resulting doses are within acceptable limits as defined in the requirements of 10 CFR Part 100, and the proposed TS changes are therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the Surveillance Requirements and Bases. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 29636). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter from O.J. Zerinque, Vice President BFN, TVA, to NRC "Browns Ferry Nuclear Plant (BFN) - Technical Specification (TS) No. 322," dated March 25, 1993.

2. Letter from Pedro Salas, Manager of Site Licensing, TVA to NRC "Browns Ferry Nuclear Plant (BFN) - Technical Specification (TS) No. 322, Revision 1 - Elimination of Main Steamline Radiation Monitor Scram and Isolation Function," dated April 4, 1994.

3. General Electric Topical Report NEDO-31400 "Safety Evaluation of Eliminating the Boiling Water Reactor Main Steamline Isolation Valve Closure Function and Scram Function of the Main Steamline Radiation Monitor," May 1987.

4. Letter From Robert F. Janeck, BWROG, to NRC transmitting NEDO-31400, dated July 9, 1987.

5. Letter from L. Cunningham, NRC, to D. Grace, BWROG, "Resolution of Outstanding Issues Relative to Topical Report NEDO-31400," dated September 6, 1988.

6. Letter from D. Grace, BWROG, to L. Cunningham, NRC, "Response to NRC Questions on NEDO-31400, dated April 4, 1989.

7. Letter from S. Stark, GE, to L. Cunningham, NRC, "Additional Information Pertaining to NRC Review of NEDO-31400," dated August 24, 1989.

8. Letter from S. Floyd, BWROG, to L. Cunningham, NRC, "Additional Information Pertaining to NRC Review of NEDO-31400: Operator Action on MSLRM Alarm," dated October 30, 1989.

9. Letter from S. Floyd, BWROG, to L. Cunningham, NRC, "Response to Action Items from March 22, 1990 NRC/BWROG Meeting," dated April 19, 1990.

10. Letter from A. Thadani, NRC, to G.J. Beck, BWROG, "Acceptance for Referencing of Licensing Topical Report NEDO-31400," dated May 15, 1991.

11. Browns Ferry Alarm Response, Panel 9-3, page 8, Maim Steamline Radiation High, Unit 2.

12. BFN Surveillance Instruction 0-SI-4.8.B.1.a.1 Airborne Effluent Release Rate, Revision 23, dated November 17, 1993.

13. BFN Chemical Instruction CI-705 Fuel Performance and Isotopic Trends, Revision 7, dated August 10, 1993.

14. Policy Issue for The Commissioners from James M. Taylor, EDO, SECY-93-067, "Final Policy Statement on Technical Specifications Improvements," dated March 17, 1993.

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